TXU Power

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Ref: #10CFR50.73(a)(2)(iv)(A)

CPSES-200602238
Log # TXX-06184

December 18, 2006

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-446
ACTUATION OF REACTOR PROTECTION SYSTEM
LICENSEE EVENT REPORT 446/06-002-00

Gentlemen:

Enclosed is Licensee Event Report (LER) 06-002-00 for Comanche Peak Steam Electric Station Unit 2, "Reactor Trip Due to a Secondary Transient Initiated During Load Rejection Testing."

This communication contains the following new licensing basis commitments regarding CPSES Units 1 and 2:

Commitment No.	Description
27416	Operating procedures will be reviewed related to the sequencing of secondary pumps to ensure the MFW pump steam control valve remains in an effective throttling range.
27417	Training will be developed on low power events to ensure that lessons learned from this event are shared.
27418	Secondary system controller responses for Main Steam indicated flow and changes in the dampening for control inputs in the secondary system will be evaluated.

The commitment number is used by TXU Generation Company LP for the internal tracking of CPSES commitments.

IE22

Sincerely,

TXU Generation Company LP

By: TXU Generation Management Company LLC
Its General Partner

Mike Blevins

By: Fred W. Madden
Director, Oversight and Regulatory Affairs

GLM
Attachment

c - B. S. Mallett, Region IV
M. C. Thadani, NRR
Resident Inspectors, CPSES

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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Title (4)

Reactor Trip Due to a Secondary Transient Initiated During Load Rejection Testing

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Name	Docket Numbers
									N/A	05000
10	27	2006	2006	002	00	12	18	06		05000

Operating Mode (9): 1

Power Level (10): 28%

This report is submitted pursuant to the requirements of 10 CFR : (Check all that apply) (11)

	20.2201(b)		20.2203(a)(3)(i)		50.73(a)(2)(i)(C)		50.73(a)(2)(vii)
	20.2201(d)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(A)
	20.2203(a)(1)		20.2203(a)(4)		50.73(a)(2)(ii)(B)		50.73(a)(2)(viii)(B)
	20.2203(a)(2)(i)		50.36(c)(2)(i)(A)		50.73(a)(2)(iii)		50.73(a)(2)(ix)(A)
	20.2203(a)(2)(ii)		50.36(c)(1)(ii)(A)	X	50.73(a)(2)(iv)(A)		50.72(a)(2)(x)
	20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(v)(A)		73.71(a)(4)
	20.2203(a)(2)(iv)		50.46(a)(3)(ii)		50.73(a)(2)(v)(B)		73.71(a)(5)
	20.2203(a)(2)(v)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(C)		OTHER
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(B)		50.73(a)(2)(v)(D)		Specify in Abstract below or in NRC Form 366A

Licensee Contact For This LER (12)

Name	Telephone Number (Include Area Code)
Tim Hope – Regulatory Performance Manager	(254) 897-6370

Complete One Line For Each Component Failure Described in This Report (13)

Cause	System	Component	Manufacturer	Reportable To EPIX	Cause	System	Component	Manufacturer	Reportable To EPIX

Supplemental Report Expected (14)		Expected Submission Date (15)	Month	Day	Year
YES (If YES, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On October 27, 2006, Comanche Peak Steam Electric Station (CPSES) Unit 2 was in Mode 1 operating at approximately 28% power following the completion of the ninth refueling outage. During turbine load reject testing, a transient was initiated in the secondary system that resulted in a Steam Generator 2-02 Hi Hi level before manual control could be achieved. This resulted in the generation of a P14 signal causing a turbine trip and a Main Feedwater (MFW) pump trip. A manual reactor trip was initiated due to a loss of MFW. Auxiliary Feedwater automatically started and all systems responded normally during and following the trip.

The cause of this event was the initiation of an oscillation in the Main Steam system while implementing a load rejection test which caused indicated steam flow to oscillate between 0 and 1.4Mlb/HR. The MFW, Heater Drain, and Steam Dump control systems were not able to dampen the oscillations. This caused steam flow and feed flow fluctuations that resulted in a turbine trip, MFW pump trip, and subsequently a manual reactor trip. Corrective actions include evaluating changes in the dampening for control inputs in the secondary system and training to ensure lessoned learned from this event are shared.

All times in this report are approximate and Central Time unless noted otherwise.

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NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

10CFR50.73(a)(2)(iv)(A); "Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B)."

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On October 27, 2006, Comanche Peak Steam Electric Station (CPSES) Unit 2 was in Mode 1, operating at 28% power following completion of the ninth refueling outage.

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

There were no inoperable structures, systems, or components that contributed directly to the event.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE TIMES

On October 27, 2006 Comanche Peak Steam Electric Station (CPSES) Unit 2 was in Mode 1 operating at approximately 28% power following the completion of the ninth refueling outage. At 0308 hours, Operational Acceptance Testing (OAT) for a digital controls modification was being implemented. This test required four 25MWe load rejections to be induced on the Unit while at approximately 28% power. For the OAT, Operators (utility, licensed) first ensured that the steam dump pressure control [EIIS: (SB)(PC)] setpoint was 30 psig above the current Main Steam header pressure. After the control switch was placed in automatic, Operators then changed the steam dump pressure control setpoint to the current Main Steam header pressure and verified that the steam dump valves [EIIS: (SB)(PCV)] remained closed. This provided the initial conditions for the load rejection test such that the steam dump pressure control setpoint is set near the current Main Steam header pressure.

During the first two load reject tests, the steam dump valves responded as anticipated. Between the second and third load reject tests, the control rods were stepped out in a slow and deliberate manner per procedures for a total of 12 steps and Heater Drain pump [EIIS: (SN)(P)] forward flow was established in automatic to maintain Tave at zero degrees deviation to support the Reactor Coolant System (RCS) [EIIS: (AB)] leak rate test that was in progress.

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Operations considered the impact of these changes on plant parameters and determined that the resultant increase in Tave of about one degree should have no impact on the load reject testing. The aggregate impact of these plant changes was continuously evaluated. Operations was in Outage staffing and had additional resources to support the activities in process. In addition, forward flow of the Heater Drain Pumps was established after the second load rejection per normal startup procedures.

After initiation of the third load reject test, the alarm "FWPT A/B Digital CNTLR TRBL" was received. The steam dump valves and steam flow began cycling, and Main Feedwater (MFW) flow began oscillating in response to the changing steam flows. Operators closed the MFW pump recirculation valve [EIIS: (SJ)(P)(V)] (which was reported to be cycling) and placed the 2-02 MFW pump master controller [EIIS: (SJ)(P)(PMC)] in manual. When the master controller was taken to manual it was apparently at a high peak; thus, the manual setting was about 600 rpm higher than pre-test levels, increasing feed pressure over steam pressure. Operators placed the four MFW flow control valves [EIIS: (SJ)(FCV)] in manual but steam pressure and MFW flow continued to oscillate. Heater Drain pump discharge flow oscillations most likely provided the forcing function for the MFW flow oscillations during the last 60 seconds prior to the turbine trip. Steam Generator 2-02 [EIIS: (SB)(SG)] level reached the P14 "Steam Generator Hi-Hi" setpoint which resulted in a Main Turbine and MFW pump trip. The steam generator water level Hi-Hi setpoint was only exceeded for 0.65 seconds, and the P-14 did not lock in. At 0308 hours a manual reactor trip was initiated due to the loss of MFW. All control rods fully inserted, all Auxiliary Feedwater (AFW) system pumps automatically started as designed on a loss of MFW, and the unit was stabilized in Mode 3.

E. THE METHOD OF DISCOVERY OF EACH COMPONENT OR SYSTEM FAILURE, OR PROCEDURAL OR PERSONNEL ERROR

Operators in the Unit 2 Control Room received a Feedwater Pump Turbine (FWPT) A/B Digital Controller Trouble alarm.

II. COMPONENT OR SYSTEM FAILURES

A. FAILURE MODE, MECHANISM, AND EFFECT OF EACH FAILED COMPONENT

Not applicable – there were no component failures associated with this event.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

Not applicable – there were no component failures associated with this event.

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C. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURE OF COMPONENTS WITH MULTIPLE FUNCTIONS

Not applicable - there were no component failures associated with this event.

D. FAILED COMPONENT INFORMATION

Not applicable - there were no component failures associated with this event.

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

Both Motor Driven AFW pumps and the Turbine Driven AFW pump automatically started as designed.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

Not applicable – there was no safety system train inoperability that resulted from this event.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

This event is bounded by the accident analysis in Sections 15.1.2, “Feedwater System Malfunctions That Result in an Increase in Feedwater Flow” and Section 15.2.7, “Loss of Normal Feedwater Flow.” A loss of normal feedwater resulting from pump failure, valve malfunction, or loss of offsite power leads to a reduction in the capability of the secondary system to remove heat generated in the reactor core. These events are analyzed in section 15.2.7 of the CPSES Updated Final Safety Analysis Report (UFSAR) which uses conservative assumptions in the analysis to minimize the energy removal capability of the AFW system. The October 27, 2006 event occurred with the reactor at approximately 28% power. All systems and components functioned as designed. The event is bounded by the UFSAR accident analysis which assumes an initial power level of 102% and the worst single failure in the AFW system for a loss of feedwater event. The UFSAR analysis shows that a loss of normal feedwater does not adversely affect the core, the reactor coolant systems, or the steam system; therefore, this event posed no threat to the health and safety of the public.

Based on the above, it is concluded that the health and safety of the public was unaffected by this condition and this event has been evaluated to not meet the definition of a safety system functional failure per 10CFR50.73(a)(2)(v).

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IV. CAUSE OF THE EVENT

The cause of this event was the initiation of an oscillation in the Main Steam system while implementing a load rejection test which caused indicated steam flow to oscillate between 0 and 1.4Mlb/HR. The MFW, Heater Drain, and Steam Dump control systems were not able to dampen the oscillations. This caused steam flow and feed flow fluctuations that resulted in a turbine trip, MFW pump trip, and subsequently a manual reactor trip. The collective effect of the tests that were underway and the responsiveness of the secondary control systems for the specific plant conditions at this power level were not fully understood. As a result, the plant's inability to dampen the transient was not anticipated.

Contributing factors for this event included Steam Dump Valve cycling which resulted in indicated steam flow oscillating between 0 and 1.4 Mlb/HR. Forward feed from the Heater Drain pumps resulted in higher MFP suction pressure (less work for the MFW pump) which placed the MFP steam control valve nearer the less stable region so that when Feedwater header pressure began to oscillate, the MFP also began to oscillate. Plant conditions which established a slightly higher Tave for the third test with a slightly higher Main Steam pressure and larger steam dump demand may have contributed to the Steam Dump valve oscillation. Implementation of a revised gain setting on the MFP master controller prior to the outage to improve MFP speed control at 100% power may have reduced the capability to dampen oscillations at low power. Due to the Feedwater pump speed oscillations when the master controller was taken to manual, it was apparently at a high peak thus the manual setting was about 600 rpm higher than pre-test levels, increasing feed pressure over steam pressure.

V. CORRECTIVE ACTIONS

Based on a review of Unit 1 data, the remaining low power load rejection tests and the four load swings scheduled at high power were cancelled. The review determined that no additional information was necessary for digital system performance enhancements. Operations performed a normal startup and brought the Heater Drain system on at a higher power level of about 40%.

The gain on the MFW pump master controller was restored to its original value and the OAT was successfully completed. Operating procedures will be reviewed related to the sequencing of secondary pumps to ensure the MFW pump steam control valve remains in an effective throttling range. Secondary system controller responses for Main Steam indicated flow and changes in the dampening for control inputs in the secondary system will be evaluated. Training will be developed on low power events to ensure that lessons learned from this event are shared, and testing planned subsequent to the twelfth refueling outage on Unit 1 will be reviewed for application of lessons learned from this event.

VI. PREVIOUS SIMILAR EVENTS

There have been no previous similar reportable events at CPSES in the last three years.